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Title: Transmutation Feature Within MCNPX

Gregg W. McKinney and Holly R. Trellue
Los Alamos National Laboratory

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Transmutation Feature Within MCNPX

Gregg W. McKinney (LANL, D-5)
Holly R. Trellue (LANL, D-5)

NNSA National Nuclear Security Administration
Los Alamos

Outline

- Existing Burnup Capabilities
- MCNPX/CINDER90 Interface
- Results
- Future Work

Existing Burnup Capabilities

- **Numerous "scripts" written to link MC codes to depletion codes**
  - MOCUP (MCNP/ORIGEN2, INEL, 1995)
  - MC-REBUS (MCNP/REBUS, ANL, 1998)
  - OCTOPUS (MCNP/FISPACT, ECN NRG Netherlands, 1998)
  - MCB (MCNP/Custom, RIT Sweden, 1999)
  - MonteBurns 2 (MCNP/ORIGEN2 or CINDER90, LANL, 1999)
  - MCWO (MCNP/ORIGEN2, INEEL, 2000)
  - BURNCAL (MCNP/Custom, SNL, 2002)
  - MCODE (MCNP/ORIGEN2, MIT, 2002)
- **Disadvantages of a "link" approach**
  - Several input files to create and understand
  - Numerous input/output files to manage
  - Approximations to convert data from one format/code to another

MCNPX/CINDER90 Interface

- **MCNPX provides to CINDER90**
  - 63-group fluxes in each material to be burned
  - Isotopic atom densities and material volumes
  - Absorption and fission reaction rates for each nuclide
  - Average k_{eff} and fission ν, and fission Q
  - Power level and burn time
- **CINDER90 provides to MCNPX**
  - Updated isotopic atom densities
  - Burnup quantities
- **User interface (BURN card)**
  - BURN card without any entries defaults to 1MW power for 1 day
  - User can specify burn materials, power level, burn times, etc.
  - Histories run per burn time are taken from NPS or KCODE card

BURN Card

- **Format**

  BURN POWER=P TIME=T_1,T_2,… PFRAC=F_1,F_2,… MAT=M_1,M_2,...
  OMIT=L_1,N_1,I_{11},I_{12},…L_2,N_2,I_{21},I_{22},… AFMIN=A BOPT=B_1,B_2,B_3

- **Entries**

  P = power level (MW). Default is 1.0.

  T_i = duration of the i^{th} burn step (days). Default is one time step of one day.

  F_i = power fraction of each time step (0-1). Zero gives decay only. Default is 1.0.

  M_j = list of burn material numbers. Default is to burn all materials.

  L_k = k^{th} material for which to omit nuclides. If L_1=-1, list applies to all materials.

  N_k = number of nuclides listed for the k^{th} material.

  I_{k1},I_{k2},… = omitted nuclide list for the k^{th} material. Format is zzaaa.

  A = threshold atom fraction. Default is 1.0e-10.

  B_1 = fission Q multiplier. Default is 1.0.

BURN Card

- **Example**

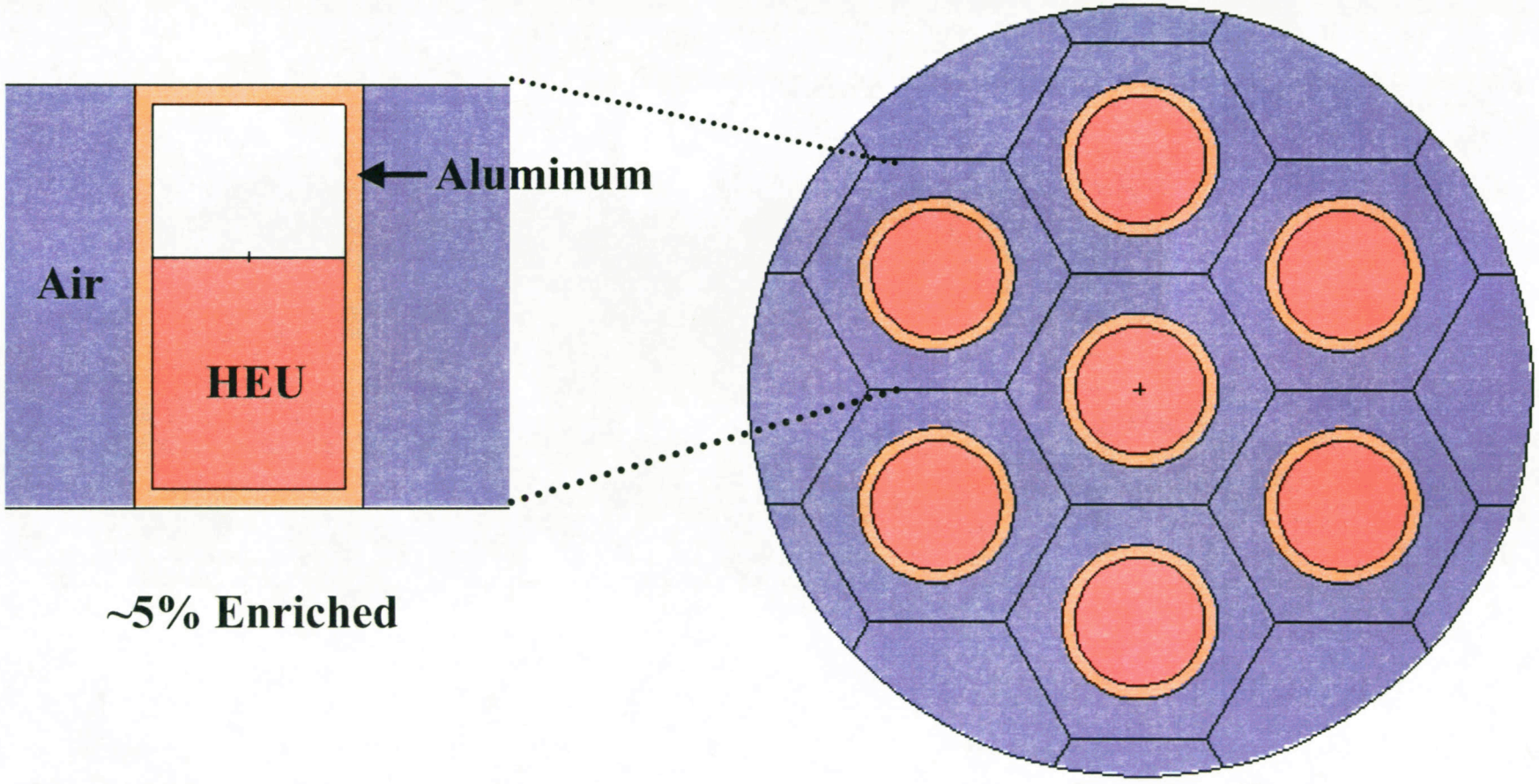
BURN POWER=2.0 TIME=15,30,30 MAT=3,4
OMIT=3,3,8017,92234,94239,4,1,92234

Specifies a power level of 2 MW for a duration of 75 days (steps of 15, 30, and 30 days). Materials 3 and 4 are included in the burn with isotopes ^{17}O, ^{234}U, and ^{239}Pu excluded from material 3 and isotope ^{234}U excluded from material 4. Nuclides with an atom fraction less than 1e-10 are also excluded. To force the inclusion of a nuclide simply list that nuclide on the appropriate material card with an insignificant atom fraction.

Results

- **7-can HEU test problem**
- **Comparison to MonteBurns**
- **Actinide and FP inventories**
- **Can burnup**

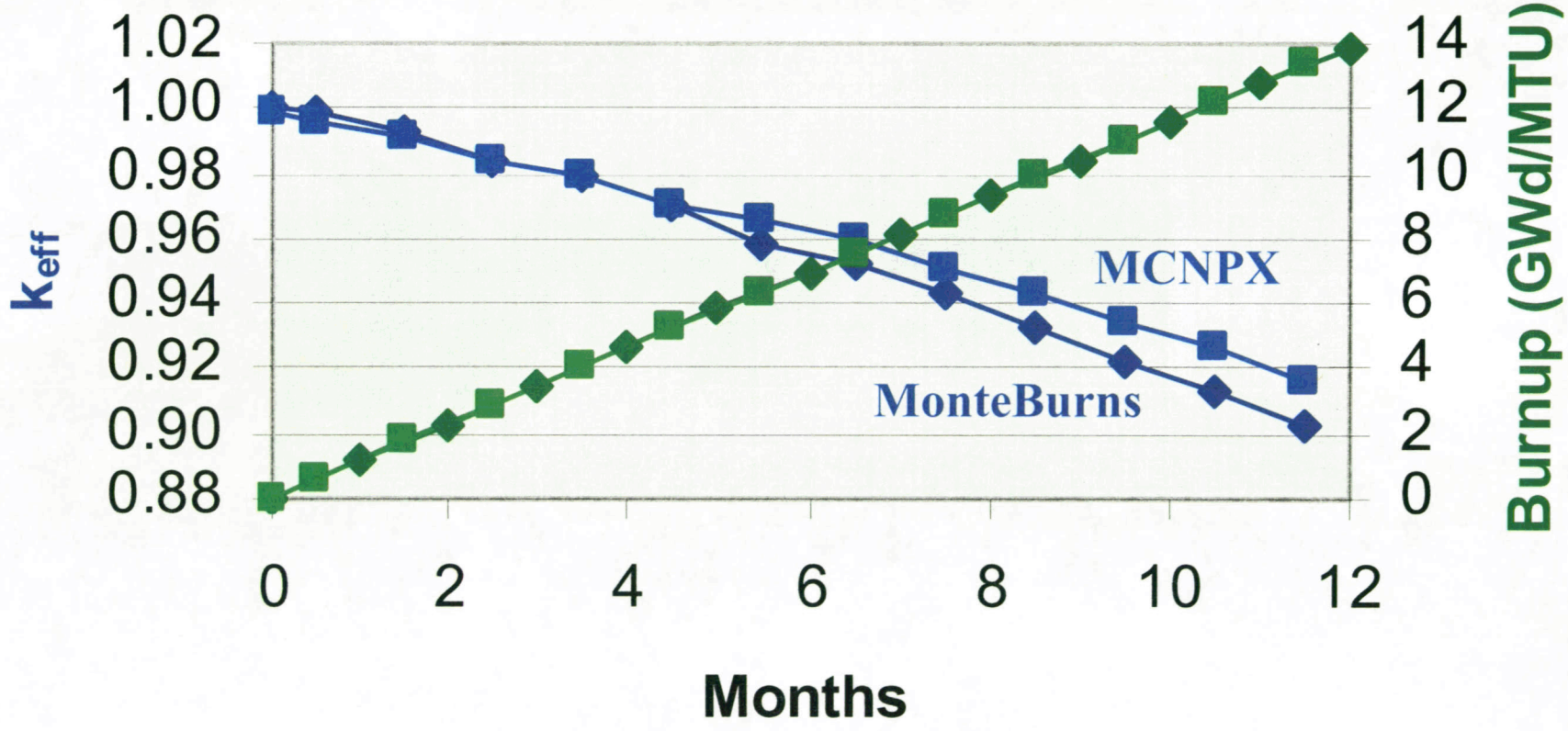
7-Can HEU Test Problem



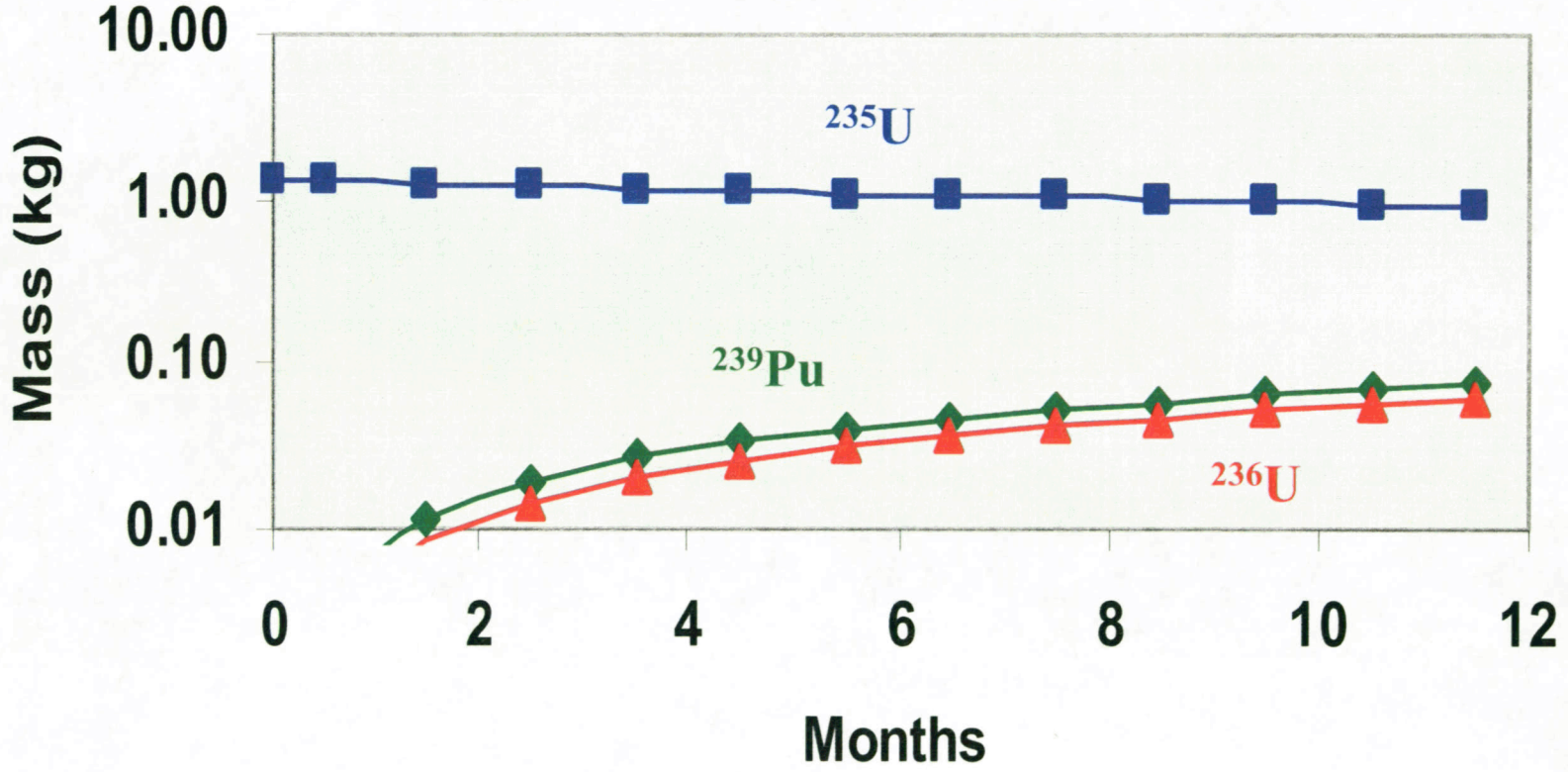
~5% Enriched

```
cylinders containing critical fluid in macrobody hex lattice
1 1 -8.4       -1       u=1    imp:n=1
2 0            -2       u=1    imp:n=1
3 2 -2.7       -3 1 2   u=1    imp:n=1
4  3 -.001      3       u=1    imp:n=1
10 3 -.001     -6 lat=2 u=2    imp:n=1 fill=-2:2 -2:2 0:0
                                        2 2 2 2 2
                                        2 2 1 1 2
                                        2 1 1 1 2
                                        2 1 1 2 2
                                        2 2 2 2 2
11 0           -8               imp:n=1 fill=2
50 0            8               imp:n=0

1 rcc 0  0 0  0 12 0  5
2 rcc 0 12 0  0 8  0  5
3 rcc 0 -1 0  0 22 0  6
6 rhp 0 -1 0  0 22 0  9 0 0
8 rcc 0 -1 0  0 22 0  30

m1     1001 5.7058e-2   8016 3.2929e-2
       92238 2.0909e-3  92235 1.0889e-4
m2    13027 1
m3    7014 .8 8016 .2
vol   6597.344573
burn   time=15.22,30.44,30.44,30.44,30.44,30.44,30.44,30.44,30.44,
       30.44,30.44,30.44   mat=1 bopt=0.99
       omit=-1,9,8017,92234,92239,93235,93236,93238,93239,94236,94237
kcode 5000 1 25 225
ksrc 0 6 0   18 6 0   -18 6 0   9 6 15   -9 6 15   9 6 -15   -9 6 -15
```

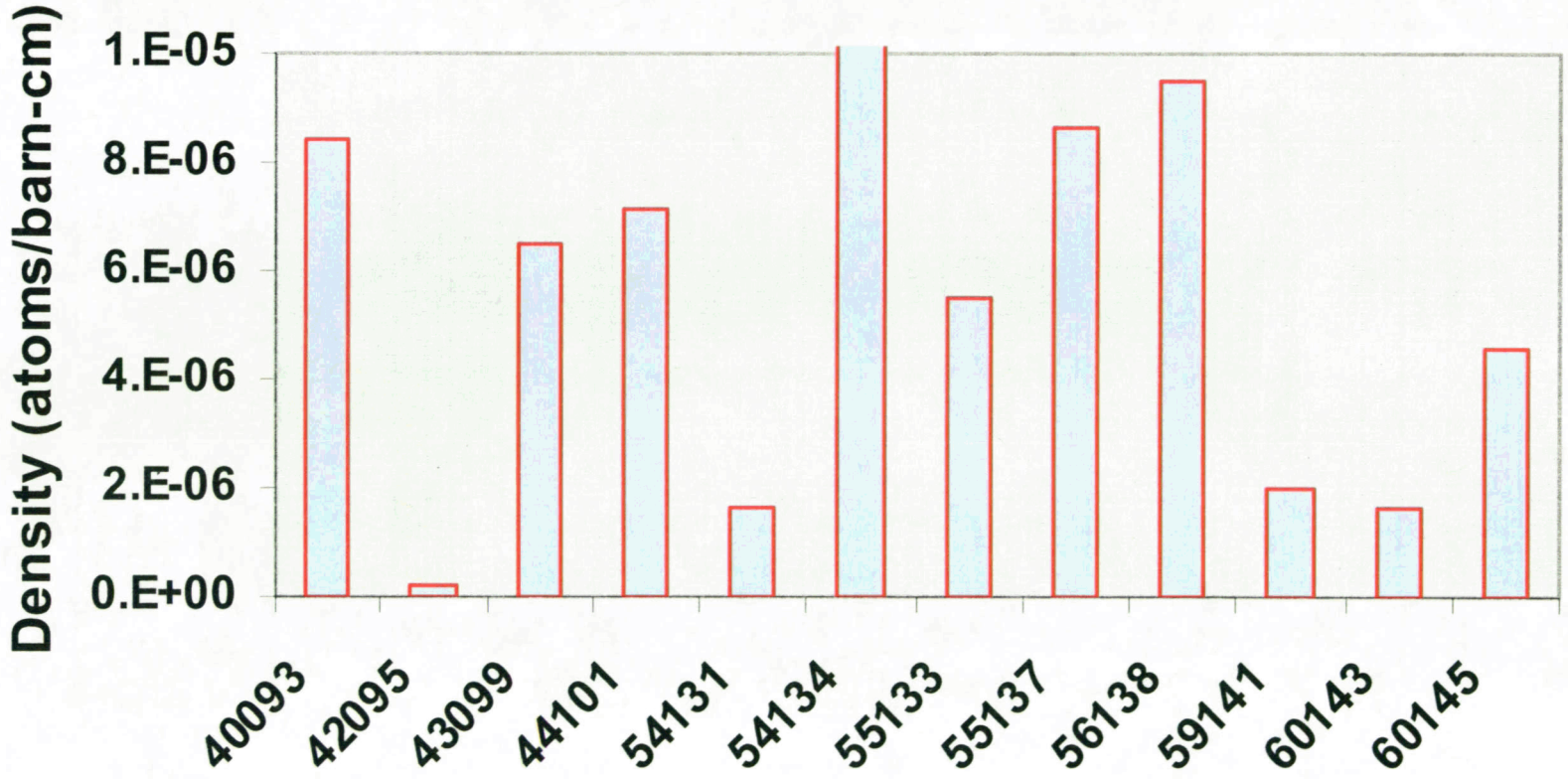
Comparison to MonteBurns



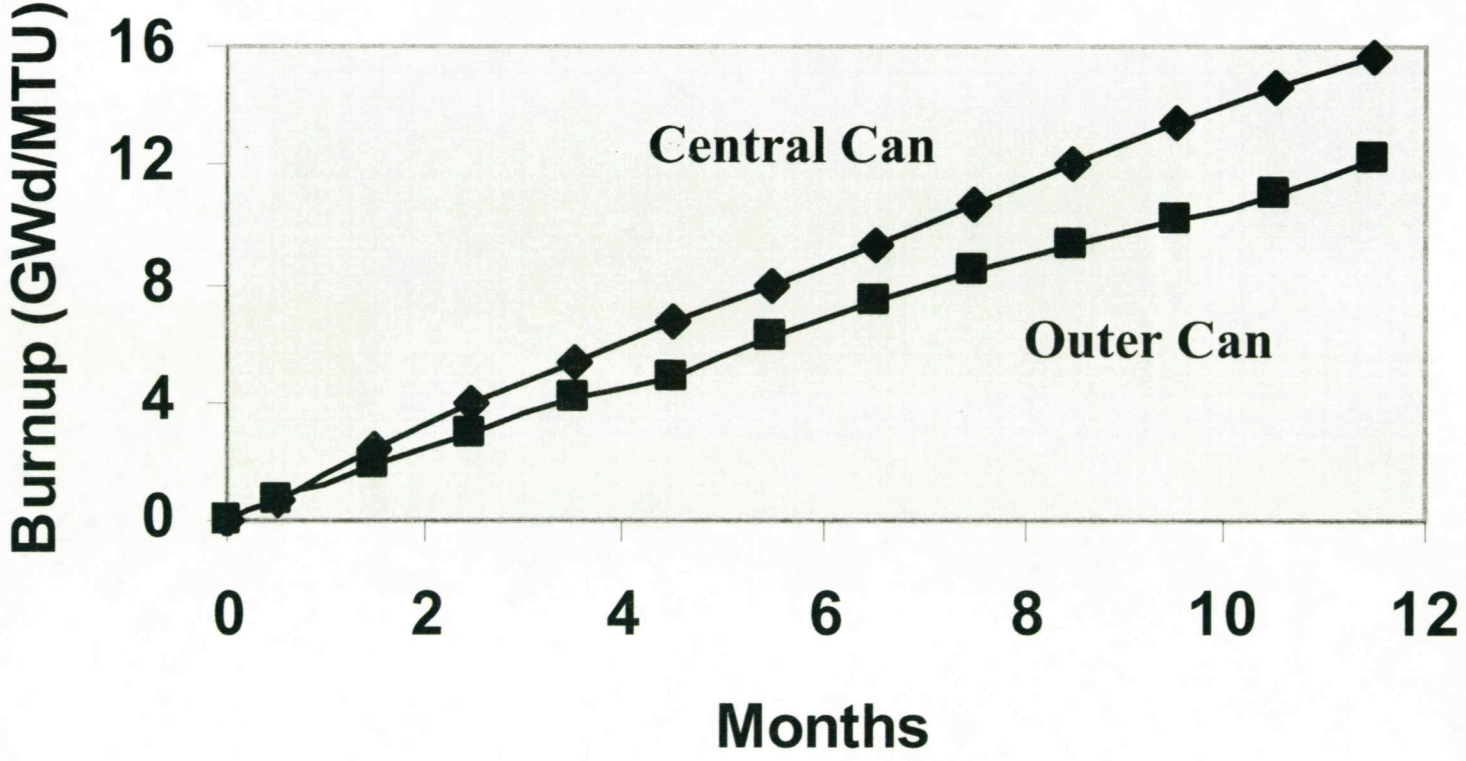
Actinide Inventories



Fission Product Inventories



Can Burnup


Burnup (GWd/MTU)
16
12
8
4
0
Central Can
Outer Can
0
2
4
6
8
10
12
Months

Future Work

- **Provide burnup tables in the MCNPX output file**
  - Densities, atom fractions, and burnup for each time step
  - Actinide and FP inventories for each time step
- **Allow transmutation with fixed-source problems**
  - Coupling with a time-dependent source
- **Benchmark with other codes & measurements**
  - Understand differences with MonteBurns
  - Benchmark with other codes (MCB, MCWO, MCODE, etc.)
  - Benchmark with measurements (ATW, MIT, IAEA)
- **Release in a future version of MCNPX (2.6.X)**
  - 2.5.E released Feb. 2004
  - 2.5.0 expected June 2004